2003 RelapS International Users Seminar West Yellowstone, Montana (US) August 27-29, 2003

Peach Bottom Turbine Trip Benchmark Analysis With RELAP5-3D@ Coupled Code

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Abstract -*The fast and recent progress in computer technologies make possible the development of codes with 3Dimensional (3D) modeling capabilities addressed to the simulation ofphenomena involving thermal-hydraulics and neutronics interaction feedback effects.*
A series of benchmark has been set up by NEA/OECD with support of NRC to challenge 3D *Neutronics-Thermal-hydraulics coupling codes performance capabilities. The first proposed benchmark was based on PWR Main Steam Line Break accident, after this experience a BWR computational problem based on well defined experimental data has been selected. The reference BWR reactor is the Peach Bottom Boiling Water Reactol:*
The present document deals with the results obtained in the analysis of the Peach Bottom Turbine Trip Test 2 *Benchmark with the* 3D *neutronics/thermal-hydraulics coupled code REIAP5-* 3De .*The core response has been evaluated and the results have been compared with REIAP5/PARCS calculations and experimental data.*

I. **INTRODUCTION**

One of the main features of 3D neutronics coupled codes is addressed to "best estimate" calculations involving transients where complex interaction between core behavior and plant dynamics occurs.
The NEA of the OECD has established a series of benchmark focused on the different LWR reactor types with the objective to verify the capabilities of the codes to perform complex analysis transient where the 3D phenomena playa role in term of best estimate calculation.
The actual trend in computer code development is focused is the assessment of codes with these capabilities.
The Peach Bottom 2 (PB2) Turbine Trip 2 (TT2) Benchmark. consists of three separate exercises: the exercise 1 tests the thermal-hydraulic system response and initializes the participants' system models (the core power response is fixed to reproduce actual test results); the exercise 2 tests and initialize the coupled 3-D kinetics/core thermal hydraulic boundary conditions (BC) models; and the exercise 3 is the best estimate coupled 3-D core! thermal hydraulic system modeling. The work presented in

this paper focuses on the calculations for the third exercise.

The cross-section library was generated by the benchmark team, in order to remove the uncertainties introduced with using different cross-section generation and modeling procedures.
Application of RELAP5-30@} to the Benchmark allows investigating the capability to adopt the symmetry option of the code.
This paper provides a description of the model developed for RELAP-30 [I] starting from the one assessed for RELAP5-PARCS [2] and also it presents the comparison between RELAP5-30@} and RELAP5/PARCS results obtained by performing the PB2 TT2 Benchmark.

II. BENCHMARK DESCRIPTION

The reactor taken as reference is the Peach Bottom, unit 2, GE-designed BWR/4 and it is based on the information provided in the EPRI reports [3.4].
Nineteen assembly types are contained within the core geometry. There are 435 compositions. The corresponding sets of cross-sections are provided. Each composition is

defined by material properties (due to changes in the fuel design) and burn-up. The burn-up dependence is a three-component vector of variables: exposure (GWd/t), spectral history (void fraction) and control rod history.
A complete set of diffusion coefficients, macroscopic cross-sections for scattering, absorption, and fission, assembly discontinuity factors (ADFs), as a function of the moderator density and fuel temperature is defined for each composition. The group inverse neutron velocities are also provided for each composition. Dependence of the cross- sections on the above variables is specified through a two-dimensional table look-up [4].
The transient begins with a sudden Turbine Stop Valve (TSV) closure (0.1 s). The pressure oscillation generated in the main steam piping propagates with relatively little attenuation into the reactor core. The induced core pressure oscillation results in dramatic changes of the core void distribution and fluid flow. The magnitude of the neutron flux transient taking place in the BWR core is strongly affected by the initial rate of pressure rise caused by pressure oscillation and has a strong spatial variation. The correct simulation of the power response to the pressure pulse and subsequent void collapse requires a 3-D core modeling. The void collapse produces a power peak equal to 280% of nominal rated power, and occurs after about 0.75 s from the beginning of the transient.
The Scram takes place at 0.75 s from the beginning of the transient, that is basically at the time of the power peak.

III. **NODALIZATIONS**

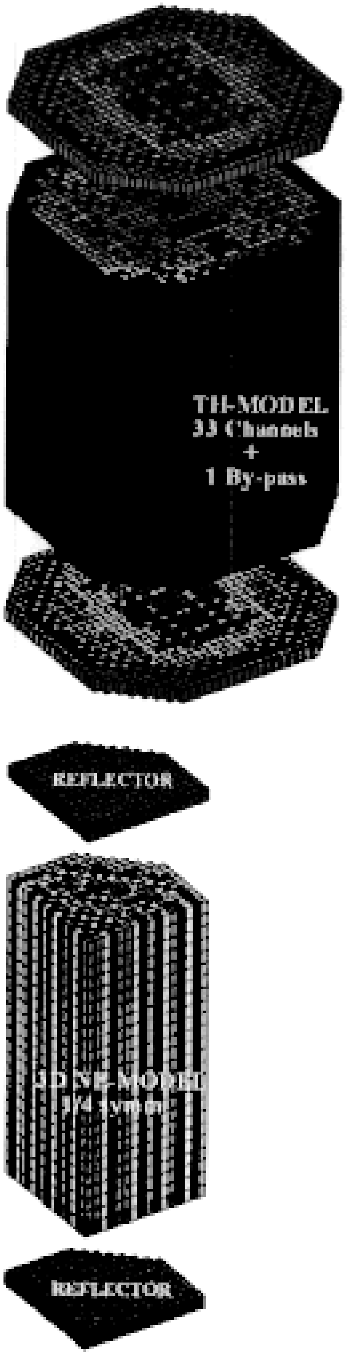
III.A. Thermal-hydraulics Nodalization

The thermal-hydraulics nodalization adopted is directly derived from the thermal-hydraulics input deck of previous RELAP5-PARCS calculations [5]. Any tuning on pressure loss coefficients has been operated on the RELAP5-3DQ input deck. The plant has been entirely modeled and particular attention has been paid in the core nodalization model due to the 3D characteristics of the phenomena involved. The core is represented by 33 parallel heated channels in order to reproduce a 3D fictitious region with *114* radial symmetry structure and the by-pass channels are modeled with one single channel. A description of the thermal-hydraulics nodalizations is reported in Ref. [6]. The four loops are lumped into one and the 2 sets of 10 jet pumps are modeled with 2 jet pump components of RELAP5 code.
The thermal-hydraulics nodalization has been considered qualified from previous RELAP-PARCS calculations.

111.8. 3D Neutronic Nodalization

The core is modeled with 26 axial slabs uniformly subdivided and grouping the 33x33 radial meshes into 19 non-adjacent zones. The criteria adopted for the grouping takes into account fuel properties, burn up and thermal- hydraulics characterization of the assemblies. The fuel is characterized with a set of 435 compositions. The 3D neutronics nodalization of the core has been developed considering the 1/4 "quasi"-symmetry of the core compositions structure. Since, the radial spatial disposition of the compositions provided by the benchmark is not exactly symmetric, in order to use the symmetry option of the RELAP5-3Dc , the compositions placed on the boundary nodes of the radial symmetry axes have been slightly modified.

Considering the present analysis a preliminary calculation and comparing with RELAP/PARCS calculations, any discontinuity factors between the neutronic nodes and any Xenon corrections have been considered in the RELAP5-3Dc runs.



F **Figure** 1 **-Coupling Scheme between 3D neutronics and Thermal-hydraulics.**

I/I.C. 3D Coupled Structure

The coupling between the thermal-hydraulics nodalization and the neutronic nodes has been built according to the scheme shown in Figure I. The core region is thermal-hydraulically characterized with 33+ 1

channels axially subdivided into 26 nodes consistent with the axial neutronic meshes. The bottom and upper slabs represent the reflector regions where two neutronic compositions cover 6 inlet and outlet volumes of the core respectively. The core bypass channel is associated with the radial reflector composition.

The 3D geometric correspondence between neutronics and thermal-hydraulics nodes assures the coupling among hydraulic volumes. heat structures and compositions. Since the number of compositions of the full radial core is lower than the number of channels in a radial cut, the same composition can characterize different channels.

IV. RESULTS

As aforementioned, the base case is obtained adopting the same thermal-hydraulics input deck used for RELAP5/PARCS calculations and developing a 3D neutronics nodalization using the lA symmetry option of RELAP5-3DQ.

The procedure followed with RELAP5-3DQ code for the set up of a 3D coupled calculation consists essentially into 2 steps, the steady state phase and the transient phase.

Due to the fact that the 3D neutronic input deck is coupled with the thermal-hydraulics one from the beginning of the calculation, the steady state condition is obtained enabling the steady state option of the code for lOOs in order to avoid any feedback oscillation. The table 1 contains the comparison among the experimental data and calculated ones at steady state. Then as second step, a restart of the same input with the transient option is run for 5 seconds.

Parameter

Core thermal Power, **MWt**

FW **mass flow**
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FW **temperature,**
K

Reactor pres~

Core Mass flow, -kg/s

Core press drop
measured/calc, Pa

Specificatio
os

2030

980.26

442.31

6798470

10445

83567.41 113560.7

RELAP
5-3Dc

2030

980.26

444.5

I **6802330J**

9382

83500

RELAPSI
PARCS

2026

980.26

444.5

6824000

99490

Table 1 **-Steady State Comparison data.**

The time trend of the main parameters is plotted in Figures 2 to 5. The comparison between RELAP5-3De , RELAP5/PARCS and experimental data is also reported in the previous mentioned figures.

Since has not been perfonned any optimization of the thennal-hydraulics and the 3D neutronics the results can be considered as preliminary.
The peak power prediction (Figure 2) is in accordance with the RELAP5/PARCS calculation, and it occurs at 0.76s and its value is slightly lower than the experimental one.

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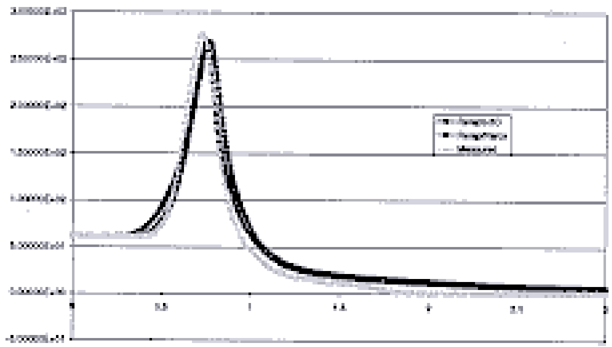

T-I" Figure 2 **-Reactor Power Comparison.**

The Pressure dome time trend (Figure 3) agrees with the calculated data of RELAP5/PARCS analysis mostly in the fIrst instants of the transient, then it differs in peak pressure wave value and stays higher till 5 seconds.

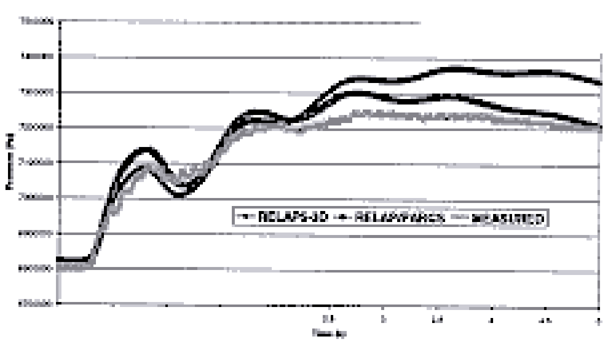

Figure 3 **-Steam Dome Pressure Comparison.**

The pressure drop across the core calculated by the 2 codes has been compared (Figure 5). The two curves have the same time trend and if superposed they practically coincide. **In** Figure 4 the pressure at core inlet and outlet is compared: a small difference is observed at the beginning of the transient then RELAP5-30c calculation predicts higher values. The most relevant evidence is that the code versions implement different interfacial drag models, and this plays a role in those transients where two-phase phenomena are involved.

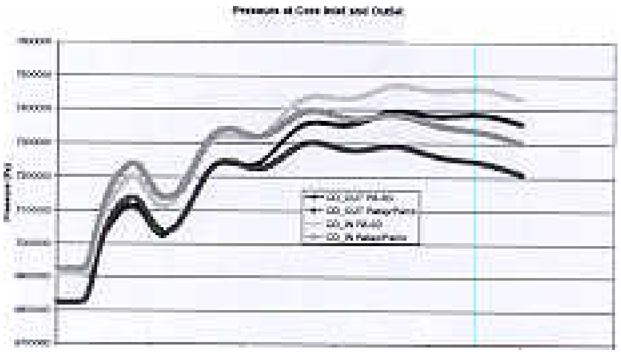

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Figure 4 **-Core Inlet and Outlet Pressures Comparison.**

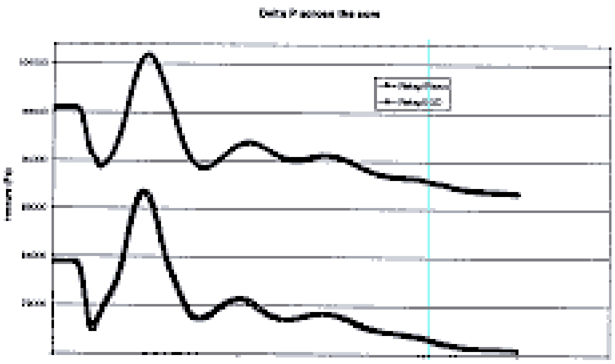

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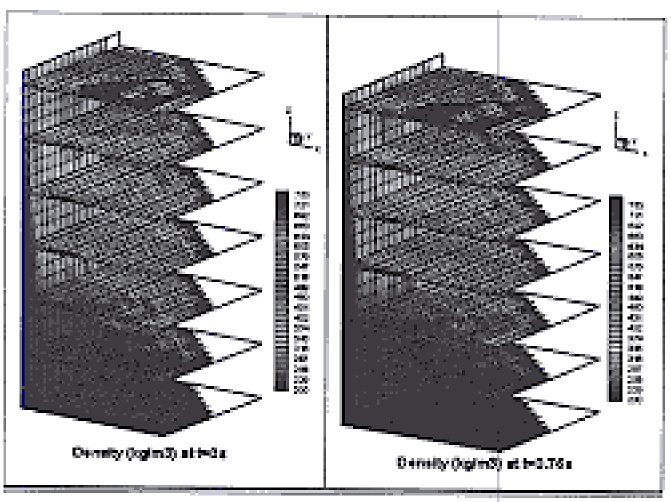
Figure 5 **-Pressure Drop Comparison across the core.**

From Figures 6 to 8, the spatial distribution of density, power and relative power in the core is shown for 2two different instants, namely at steady state and when power peak occurs.



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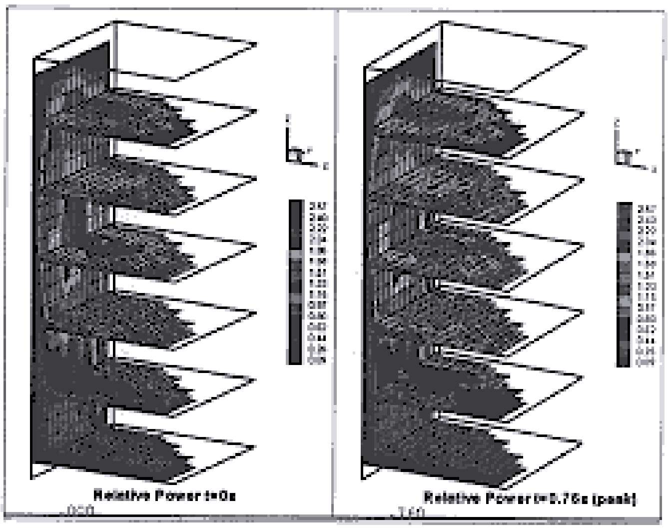
<u>Fil!ure</u> <u>7</u> **<u>-Snatia] Power Di~trihution at Steadv State and Power neak</u>**



<u>Fil!ure</u> <u>8</u> **<u>-Snatial Relative Power Di.1;trihution at Steadv State and at Power neak</u>**

When the pressure wave attains the bottom of the core and quickly propagates across the density increases and the local power rises because of the reactivity insertion due to the void feedback (Figures 6 and 7). The relative power distribution across the core does not significantly change, and the increase is mostly observed at the lower slabs.

v. **CONCLUSIONS**

The RELAP5-3D~ code has been applied to the Peach Bottom Turbine Trip Benchmark. The capabilities of the code to predict the transient scenario have been challenged and compared with experimental data and with results from RELAP/PARCS calculations. The RELAP5-3D~ lA symmetry option of core representation has been tested. An effort to adapt lA symmetry option of the RELAP5-3D~ to

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the actual PB2 geometry developed for RELAP5-PARCS has been performed.

Different interfacial drag model between RELAP5- 3De and RELAP5/PARCS codes in connection with the absence of tuning in the RELAP5-3De thermal-hydraulics nodalization development play an important role in the pressure time trend after the early pressure wave.

Concerning the Thermal-hydraulics nodalization a more accurate tuning dedicated to the RELAP5-3De input deck is needed, for the NE model, the cross section data function of the 2nd order derivative term dependence is suitable.

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[2]

[3]

[4]

[5]

[6]

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RELAP5-3D~ Code Manual, INEEL-EXT -98- 00834. February 2001.

RelapS MOD3.3 Code manual, *NUREG/CR*- SS3S/REV.l, January 2002.

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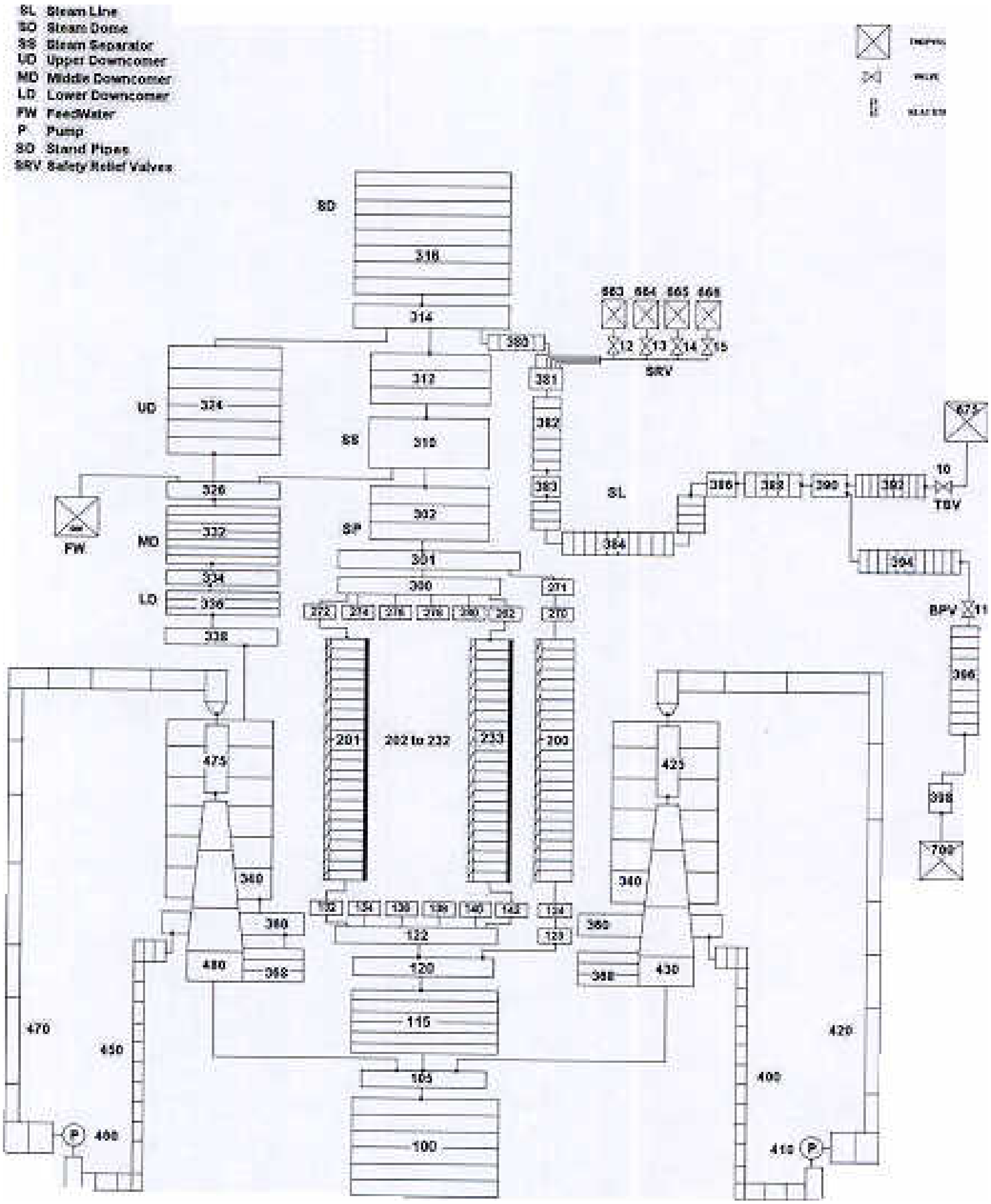

Figure 9 **-Sketch of the 33 channels Thermal-Hydraulics nodalization adopted**

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